FUEL DEPLETION ANALYSES FOR THE HEU CORE OF GHARR-1; Part II: FISSION PRODUCT INVENTORY

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ABSTRACT

The fission product isotopic inventories have been estimated for a 90.2% highly enriched uranium (HEU) fuel lattice cell of the Ghana Research Reactor-1 (GHARR-1) using the WIMSD/4 transport lattice code. The results indicate a gradual decrease in the Xe^{135} *inventory, and saturation trend for* Sm^{149}, Cs^{134} *and* Cs^{135} *inventories as the fuel is depleted to 10,000 MWd/tU.*

Introduction

In the operation of nuclear reactors, the fuel is gradually depleted or used up. In this case, the number of nuclear fission reactions occurring in the fuel in the reactor core decreases; consequently, the effective multiplication factor reduces. To compensate for reactivity loss, control rods have to be withdrawn. For the special case of GHARR-1 which has only one control rod, loss of core excess reactivity due to fuel depletion is achieved by adding beryllium shim reflectors to the top part of the core to act as upper axial reflectors.

A major contributor to fuel depletion is the production of different fission fragments or products which invariably have large absorption cross sections. Their accumulation in the reactor core tends to reduce the core reactivity. In particular, since absorption cross sections decrease rapidly with increasing neutron energy, such fission product poisons are of great importance to thermal reactor design. The amounts and activities of individual fission products and the total fission product inventory in nuclear fuel during reactor operation and shutdown have important connotations. These quantities are used to evaluate radiation hazard from fission product dispersal to the environment, to determine the fission product radioactivity in the spent nuclear fuel after discharge from the reactor core, to estimate decay heat removal and finally to calculate the poisoning or parasitic capture by fission products that accumulate during reactor operation [1]. The two most important fission products of concern in reactor core design and analysis, and also to reactor operators are Xe^{135} (a 1/v absorber) with an enormous thermal absorption cross section is 2.65b and a relatively large yield, and Sm^{149}. Other fission products, albeit of less importance are Cs^{134}, Cs^{135}. The designed core lifetime estimated for GHARR-1 is ten years if it is operated continuously at peal power of 30kW for 2.5 hours a day. The measured cold clean core excess reactivity during the zero power criticality tests is 3.99mk. This value is quite small. Therefore, to properly utilize this core excess reactivity bank in order to realize the estimated core lifetime, adequate notice and care should be taken of core reactivity consuming poisons such as fission products.

In this work, multigroup local burnup calculations have been performed on a 5-region fuel lattice cell of the 90.2% HEU GHARR-1 fuel assembly using the WIMSD/4 transport code [2] to estimate the relative concentrations of the these fission products. The results of the burnup variations of the isotopic inventories of these fission products are presented. The results would in particular be helpful in performing decay heat removal calculations for the design of a spent fuel storage cask for the HEU fuel of GHARR-1. Furthermore, the results would be used as a reference database for comparison with burnup and depletion calculations for alternative LEU fuels in accordance with the core conversion research and studies program initiated for GHARR-1, in the spirit if the global RERTR program.

Calculational Method

As it is well known, burnup calculations involve some approximation [3]. One such approximation is the decoupling of coupled effects. These effects are manifest in the microscopic-macroscopic decoupling as well as space-time decoupling. The former suggests that burnup calculations, in practice, are performed at two levels [4-8]. However, in this work, the burnup calculations have been restricted to the microscopic or lattice level. A thorough review of the theory used the WIMSD/4 lattice burnup and depletion calculations have been presented in Part I [9] of this paper. For brevity, we present only the aspects of the WIMSD calculations relevant to fission product formations.

The WIMSD burnup calculations are performed in Chain 12 and involve several steps. The third step involves the solution of the fewgroup criticality equation for the homogenized lattice cell with cross sections previously calculated in step 1 for absorption and production reactions. These are taken from the main transport (scattering) and solved at each short burnup stage using the diffusion equation

$$\left\{\Sigma_G^a + D_G B^2 + \sum_{h\neq G}\Sigma^{Gh}\right\}\Phi_G^n + \lambda\left\{\Sigma_G^{a'} + D_G B^{2'}\right\}\Phi_G^n$$
$$= \sum_{h\neq G}\left(\Sigma^{hG}\Phi_h^n\right) + \left(\rho + \lambda + \rho'\right)\chi_G \sum_h\left(\nu_h \Sigma_h^f \Phi_h^n\right) \qquad (1)$$

where the unprimed quantities are constant coefficients of the equation while those with primes denote control quantities. After solving this equation, the macroscopic fission cross sections are then computed and printed for all materials undergoing burnup and for the entire cell together with the few and two-groups fluxes according to:

$$\Sigma_{m,G}^{n} = V_{z\in m}\phi_{G,z\in m}^{n}\sum_{i=1}^{NNF}\left(N_{im}\sigma_{Gim}\right) \qquad (2)$$

The cell quantities are also calculated through summation over zones, *z*, and the two-group values over the respective groups. The fluxes are integrated over the cell volume. The fission products (***FP***) are also computed according the formula

$$FP = \sum_{G,z,m}\left\{\sum_{i}\left(N_{im}\sigma_{Gim}Y_i\right)\right\} \qquad (3)$$

The last step in the WIMS burnup calculation involves the performance of trapezoidal integration of the burnup equations to obtain new isotopic number densities for each burnable material *m* and nuclide *i*, with the burnup equation expressed in a somewhat different form as:

$$\frac{dN_i(t)}{dt} = -\lambda_i N_i(t) - A_i N_i(t) + \sum_k \delta\left(i, j_1(k)\right)\alpha_{ki}C_k N_k(t)$$

$$+\sum_k \delta\left(i, j_2(k)\right)\beta_{ki}\lambda_k N_k(t) + \sum_k \gamma_{ki}F_k N_k(t) \qquad (4)$$

where N_i = the number density of nuclide *I*

λ_I = decay constant for nuclide *i,*

A_I = absorption reaction rate for nuclide *i*

F_I = fission reaction rates

C_I = capture reaction rates

γ_{ki} = yield of fission product *i* computed from the fission of nuclide *k*

α_{ki}, β_{ki} = product fractions of isotope *k (*equal to 1 for nuclide *k* with single capture or decay product*)*

$j_1(k)$, $j_2(k)$ = identifiers of all products from isotope k and the delta functions $\delta(i,j)$ indicate that the contribution occurs when *i=j.*

As explained earlier, the first and second terms of the balance equation (Eq. 4) describe losses by radioactive decay of isotope *i*, and neutron capture respectively. The third, fourth and fifth terms represents gains due to neutron transmutation of nuclide *k* to nuclide *i,* decay of nuclide *k* to *i* and production of nuclide *i* through fission product formation respectively. Eq. 4 can be written compactly in the form

$$\frac{dN_i(t)}{dt} = -(\lambda + A_i)N_i(t) + \sum_k q'_{i,k}(t)N_k(t) = \sum_k q_{ik}(t)N_k(t) \qquad (5)$$

where

q_{ik}, are the production terms calculated from yields of fission products i from fission of nuclide k, production fractions, capture and fission reaction rates.

In particular, if $\lambda_I > \lambda_0$, the assumption $dN_i/dt=0$ is made and Eq. 4 reduces to

$$N_i(t+\delta t) = N_i(t) + \frac{\sum_k q_{ik}(t) N_i(t)}{\lambda + A_i} \qquad (6)$$

The numerical integration is completed for each cell material and all nuclides with defined half-lives less than 19 hours (the WIMS code considers such isotopes to be in equilibrium). The mode of calculations for other burnup parameters, e.g., ingredients, weight fractions, etc., is as previously described in Part I of this paper.

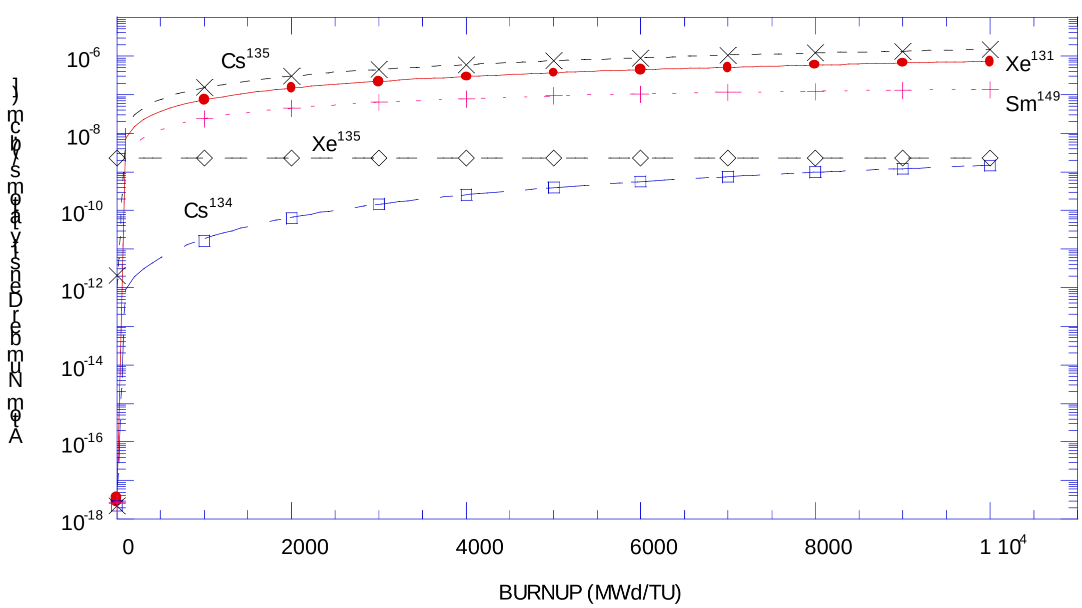
Multigroup local burnup calculations were performed for the 5-region lattice cell as previously described in Part I of this paper. All the lattice parameter results were obtained using the WIMSD/4 transport lattice code. The fuel was depleted from 0 – 10,000 MWd/tU burnup representing a maximum burnup of 1% [10] of the core. The multigroup transport equations were solved in four broad energy groups based on the WIMS 69-group library. The discrete ordinate spatial model (DSN) which solves the differential form of the transport equation by the Carlson-S_N (N=4) approach was adopted for the solution of the Boltzmann multigroup neutron transport equation.

The WIMSD/4 calculation for estimating the isotopic inventory of the fission products for the GHARR-1 fuel lattice cell is shown in Figs. 1-3. For Xe^{135}, the production is due mainly to the decay of I^{135}. A change in reactor output power and hence fuel burnup results in a change of the Xe^{135} distribution. The concentration of Xe^{135} is therefore directly proportional to the neutron flux and power density in the core. Its annihilation process occurs by neutron captures and by spontaneous decay. The shape of the Xe^{135} depletion history is therefore different from that of the other fission product isotopes. It is observed that the isotopic inventory of Xe^{135} expressed through its atom number density, ingredients and weight percent decrease with increasing fuel burnup. However, the relative change is quite small. The atom number density of Sm^{149} results from a high direct fission yield and from a preceding chain of twelve fission products of which the intermediate isotope Sm^{147} is a member. Moreover, Sm^{149} owns a high thermal absorption cross section. The Cs^{134} and Cs^{135} fission products are produced in the decay chains of Xe^{133} and Xe^{135} nuclides respectively. The variation of the relative concentrations of these fission products, estimated by their atom number densities, ingredients and weight percents with fuel burnup are also illustrated in Figs 1-3. Their isotopic inventories exhibit a saturation behavior with increasing burnup. In particular, the saturation values of the atom number densities, ingredients and weight fractions increase with fuel burnup. The rate of approach to saturation for each fission product increases as the decay constant λ_i, increases.

CONCLUSION

The isotopic inventories of some fission products relevant for the safe operation of GHARR-1 have been estimated for its 90.2% HEU fuel lattice. The WIMSD/4 transport physics lattice code based on the WIMS 69-energy group library was used to perform the multigroup lattice cell depletion analysis. The results indicate that the Xe^{135} isotopic inventory decreases gradually as the fuel is depleted. However, a build-up trend was observed for fission products Sm^{149}, Cs^{134} and Cs^{135}, attaining saturation as burnup increases to 10,100 MWd/tU, corresponding to 1% of the total core burnup.

The results would be validated with other cell depletion codes and used as input data in performing detailed 3-D reactor core depletion analysis to adequately estimate the burnup history of the GHARR-1 core, as well as in the analysis of the decay heat removal from the spent fuel storage cask, for which criticality safety and shielding analysis would be performed. Furthermore, this data would be used as reference database for comparing lattice cell parameters in the neutronic core conversion studies for the Ghana Research Reactor-1.



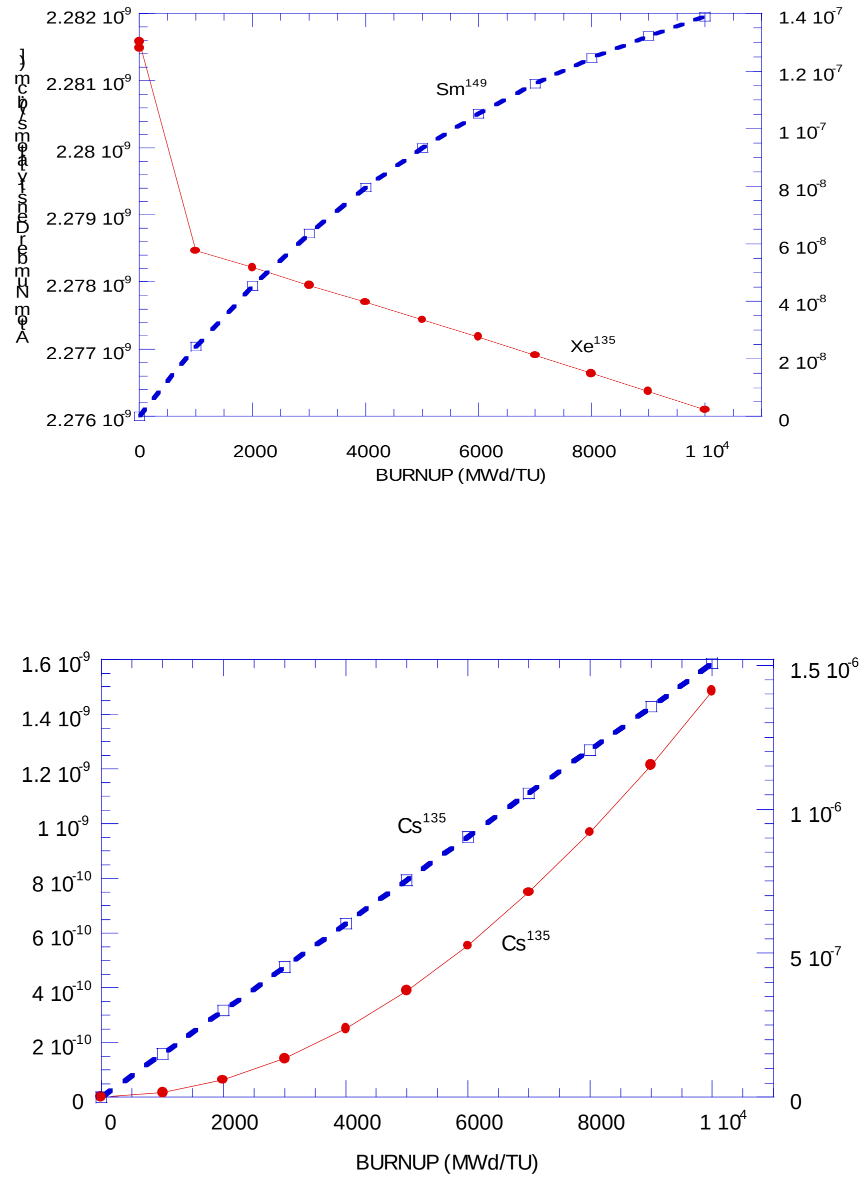


Fig. 1: Fission Product Isotopic Inventory: Atom Number Density vrs. Burnup

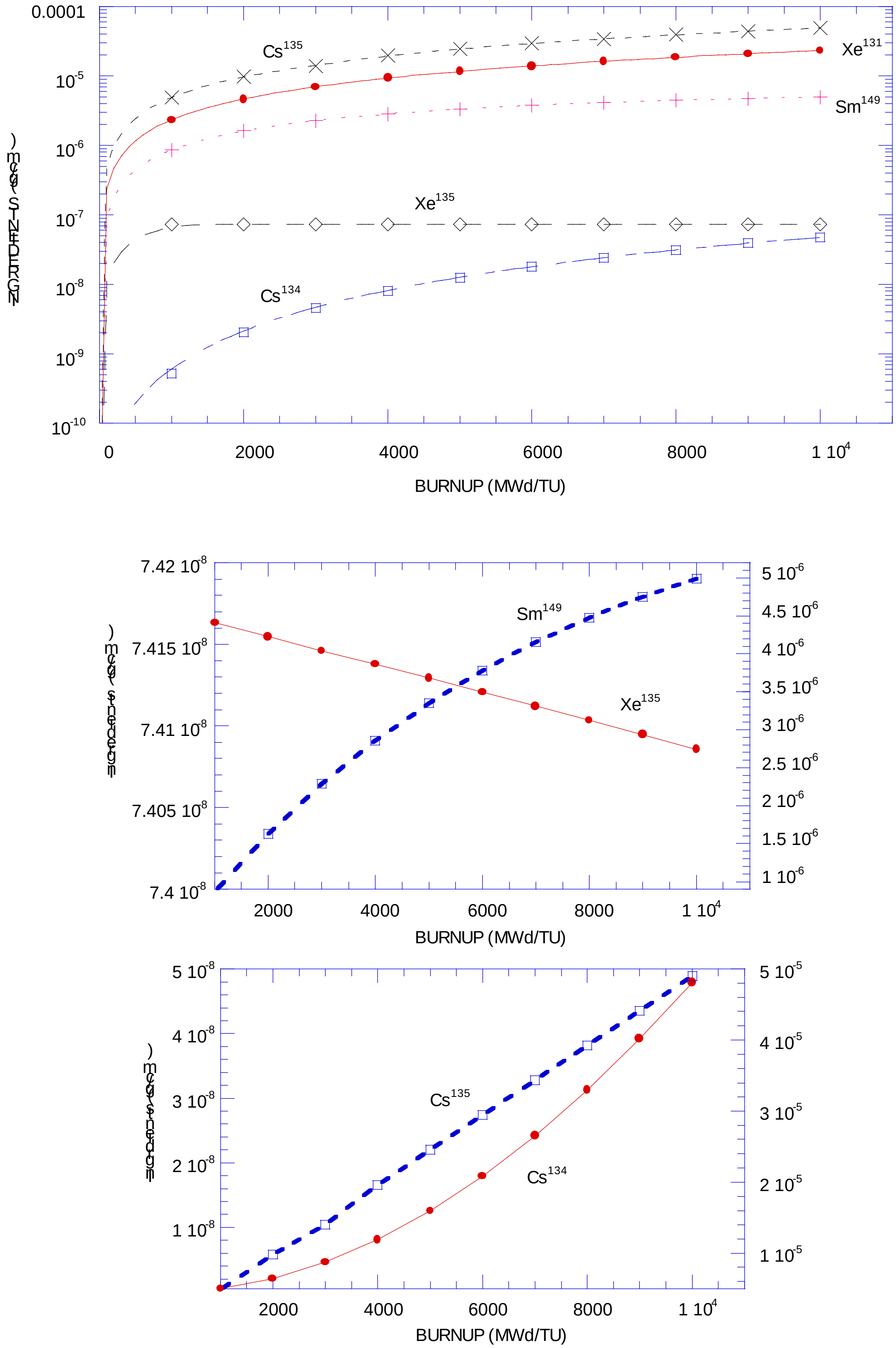


Fig. 2: Fission Product Isotopic Inventory: Ingredients vrs Burnup

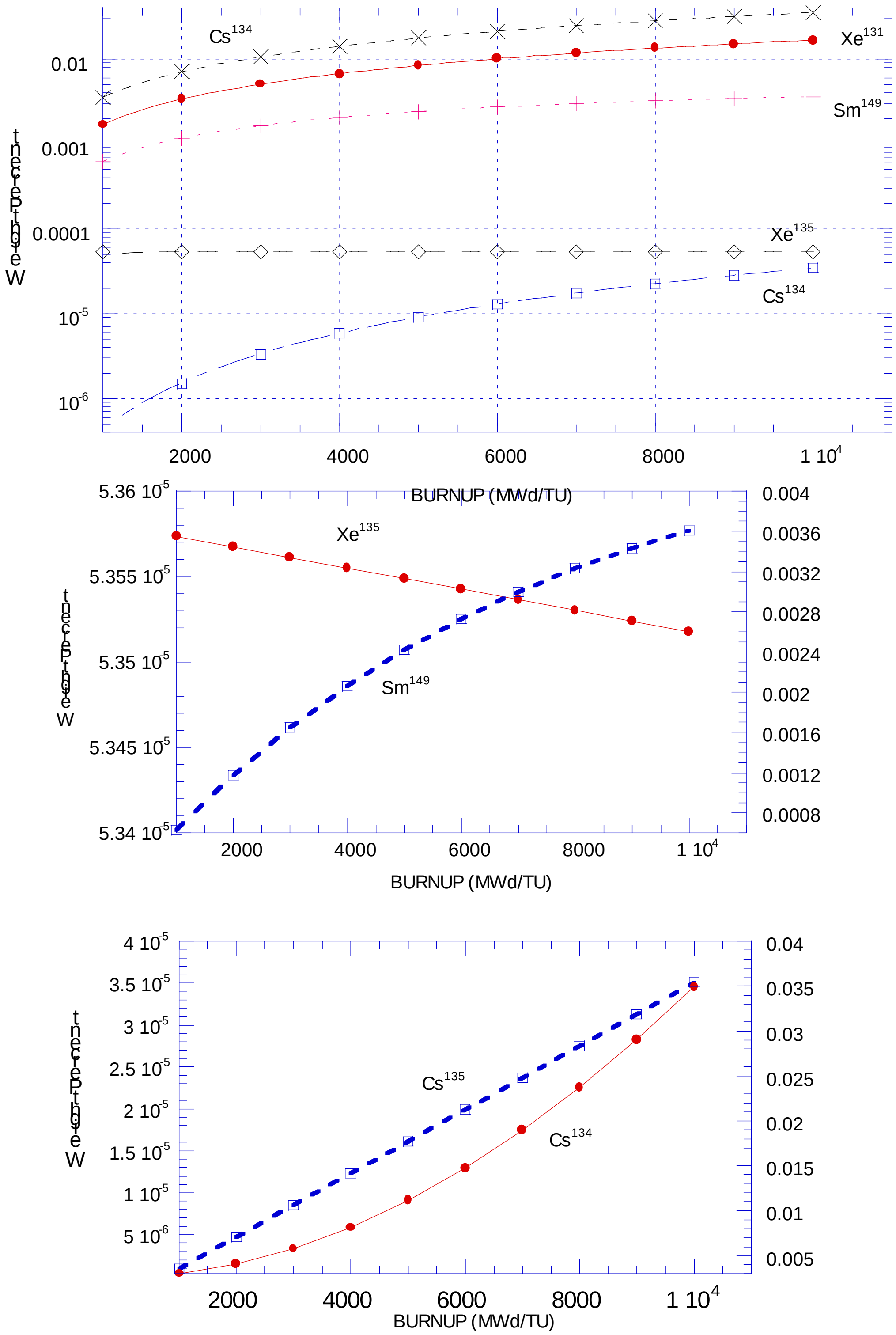


Fig. 3: Fission Product Isotopic Inventory: Weight % vrs Burnup

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